ATTACHMENT 1

EXISTING TECHNICAL SPECIFICATIONS

2.1 REACTOR CORE - Limiting Combination of Power, Pressure, and Temperature

APPLICABILITY: Applies to reactor power, system pressure, coolant temperature, and flow during operation of the plant.

OBJECTIVE: To maintain the integrity of the reactor coolant system and to prevent the release of excessive amounts of fission product activity to the coolant.

SPECIFICATION: Safety Limits

(1) The reactor coolant system pressure shall not exceed 2735 psig with fuel assemblies in the reactor.

(2) The combination of reactor power and coolant temperature shall not exceed the locus of points established for the RCS pressure in Figure 2.1.1. If the actual power and temperature is above the locus of points for the appropriate RCS pressure, the safety limit is exceeded. 60 6/8/81

Maximum Safety System Settings

The maximum safety system trip settings shall be as stated in Table 2.1. 117 12/13/88

BASIS: Safety Limits

1. Reactor Coolant System Pressure

The Reactor Coolant System serves as a barrier which prevents release of radionuclides contained in the reactor coolant to the containment atmosphere. In addition, the failure of components of the Reactor Coolant System could result in damage to the fuel and pressurization of the containment. A safety limit of 2735 psig (110% of design pressure) has been established which represents the maximum transient pressure allowable in the Reactor Coolant System under the ASME Code, Section VIII.

2. Plant Operating Transients

In order to prevent any significant amount of fission products from being released from the fuel to the reactor coolant, it is necessary to prevent clad overheating both during normal operation and while undergoing system transients. Clad overheating and potential failure could occur if the heat transfer mechanism at the clad surface departs from nucleate boiling. System parameters which affect this departure from nucleate boiling (DNB) have been correlated with experimental data to provide a means of determining the probability of DNB occurrence. The ratio of the heat flux at which DNB is expected to occur

for a given set of conditions to the actual heat flux experienced at a point is the DNB ratio and reflects the probability that DNB will actually occur.

It has been determined that under the most unfavorable conditions of power distribution expected during core lifetime and if a DNB ratio of 1.44 should exist, not more than 7 out of the total of 28,260 fuel rods would be expected to experience DNB. These conditions correspond to a reactor power of 125% of rated power. Thus, with the expected power distribution and peaking factors, no significant release of fission products to the reactor coolant system should occur at DNB ratios greater than 1.30.(1) The DNB ratio, although fundamental, is not an observable variable. For this reason, limits have been placed on reactor coolant temperature, flow, pressure, and power level, these being the observable process variables related to determination of the DNB ratio. The curves presented in Figure 2.1.1 represent loci of conditions at which a minimum DNB ratio of 1.30 or greater would occur. (1)(2)(3) 49 7/19/79

Maximum Safety System Settings

1. Pressurizer High Level and High Pressure

In the event of loss of load, the temperature and pressure of the Reactor Coolant System would increase since there would be a large and rapid reduction in the heat extracted from the Reactor Coolant System through the steam generators. The maximum settings of the pressurizer high level trip and the pressurizer high pressure trip are established to maintain the DNB ratio above 1.30 and to prevent the loss of the cushioning effect of the steam volume in the pressurizer (resulting in a solid hydraulic system) during a loss-of-load transient.(3)(4)

In the event that steam/feedflow mismatch trip cannot be credited due to single failure considerations, the pressurizer high level trip is provided. In order to meet acceptance criteria for the Loss of Main Feedwater and Feedline Break transients, the pressurizer high level trip must be set at 20.8 ft. (50%) or less. 97 4/7/86

2. Variable Low Pressure Loss of Flow and Nuclear Overpower Trips

These settings are established to accommodate the most severe transients upon which the design is based, e.g., loss of coolant flow, rod withdrawal at power, control rod ejection, inadvertent boron dilution and large load increase without exceeding the safety limits. The settings have been derived in consideration of instrument 117 12/13/88

errors and response times of all necessary equipment. Thus, these settings should prevent the release of any significant quantities of fission products to the coolant as a result of transients.(3)(4)(5)(7) | 117 12/13/88

In order to prevent significant fuel damage in the event of increased peaking factors due to an asymmetric power distribution in the core, the nuclear overpower trip setting on all channels is reduced by one percent for each percent that the asymmetry in power distribution exceeds 5%. This provision should maintain the DNB ratio above a value of 1.30 throughout design transients mentioned above. | 117 12/13/88

The response of the plant to a reduction in coolant flow while the reactor is at substantial power is a corresponding increase in reactor coolant temperature. If the increase in temperature is large enough, DNB could occur, following loss of flow.

The low flow signal is set high enough to actuate a trip in time to prevent excessively high temperatures and low enough to reflect that a loss of flow conditions exists. Since coolant loop flow is either full on or full off, any loss of flow would mean a reduction of the initial flow (100%) to zero.(3)(6)

References:

(1) Amendment No. 10 to the Final Engineering Report and Safety Analysis, Section 4, Question 3

(2) Final Engineering Report and Safety Analysis, Paragraph 3.3

(3) Final Engineering Report and Safety Analysis, Paragraph 6.2

(4) Final Engineering Report and Safety Analysis, Paragraph 10.6

(5) Final Engineering Report and Safety Analysis, Paragraph 9.2

(6) Final Engineering Report and Safety Analysis, Paragraph 10.2

(7) NIS Safety Review Report, April 1988 | 117 12/13/88

TABLE 2.1

MAXIMUM SAFETY SYSTEM SETTINGS

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	Three Reactor Coolant Pumps Operating
*1. Pressurizer High Level	≤ 20.8 ft. above bottom of pressurizer when steam/feedflow mismatch trip is not credited, or ≤ 27.3 ft. above bottom of pressurizer when steam/feedflow mismatch trip is credited
2. Pressurizer Pressure: High	≤ 2220 psig
3. Nuclear Overpower	
a. High Setting**	≤ 109% of indicated full power
b. Low Setting	≤ 25% of indicated full power
***4. Variable Low Pressure	≥ 26.15 (0.894 ΔT+T avg.) - 14341
***5. Coolant Flow	≥ 85% of indicated full loop flow

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* Credit can be taken for the steam/feedflow mismatch trip when this system is modified such that a single failure will not prevent the system from performing its safety function.

** The nuclear overpower trip is based upon a symmetrical power distribution. If an asymmetric power distribution greater than 5% should occur, the nuclear overpower trip on all channels shall be reduced one percent for each percent above 5%.

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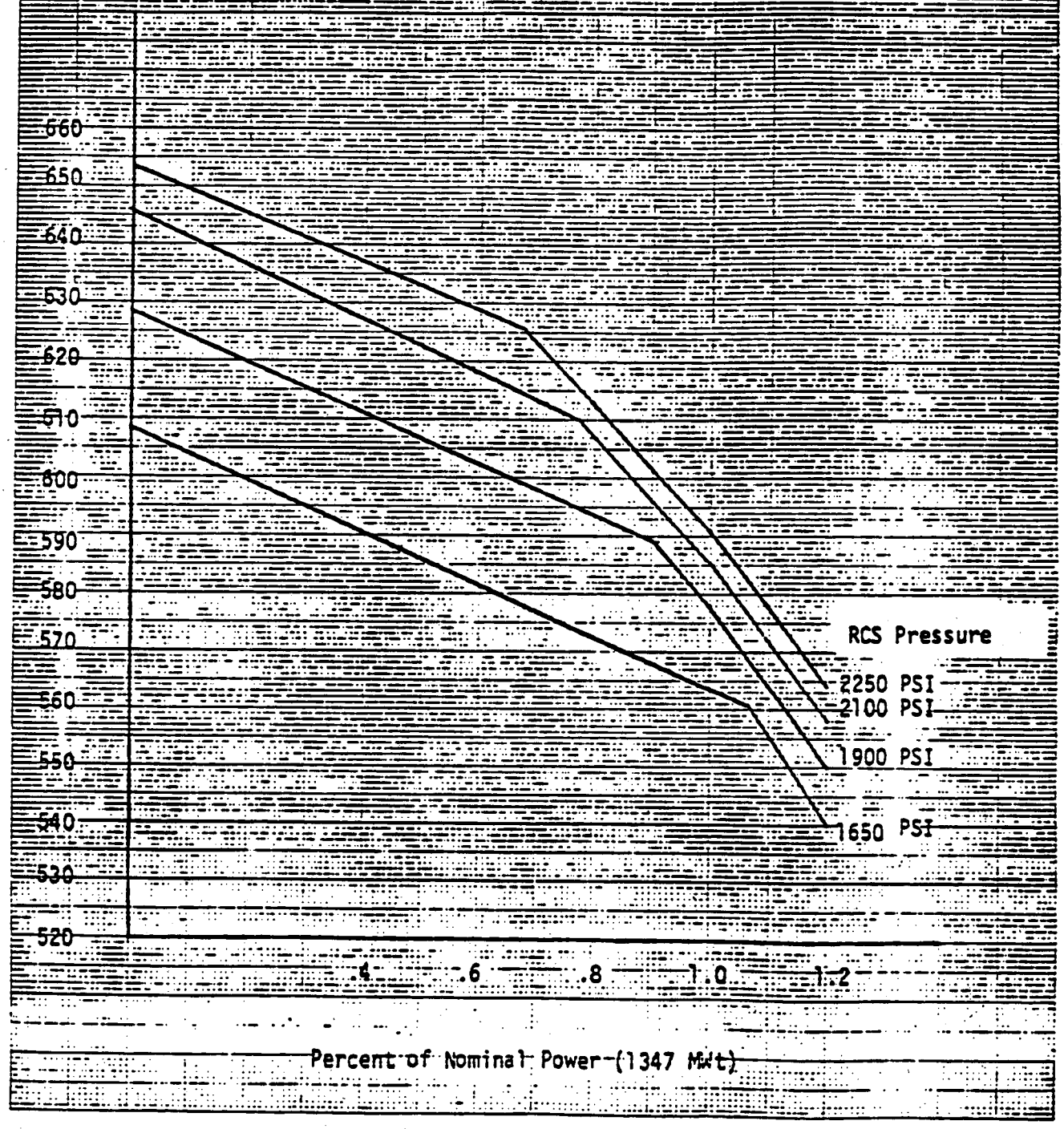
***May be bypassed at power levels below 10% of full power.

Figure 2.1.1
Safety Limits
Temperature, Power, Pressure
RCS Flow ≥ 195,000 GPM



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6/8/81

Revised: 6/23/81

3.3.3 MINIMUM WATER VOLUME AND BORON CONCENTRATION IN THE REFUELING WATER STORAGE TANK

APPLICABILITY: Applies to the inventory of borated refueling water.

OBJECTIVE: To ensure immediate availability of safety injection and containment spray water of required quality.

SPECIFICATION: When the Safety Injection System or the Containment Spray System is required to be operable, the refueling water tank shall be filled to at least elevation 50 feet with water having a boron concentration of not less than 3750 ppm and not greater than 4300 ppm. 34 4/1/77

BASIS: The refueling water storage tank serves two purposes; namely:

(1) As a reservoir of borated water for accident mitigation purposes,

(2) As a reservoir of borated water for flooding the refueling cavity during refueling.

Approximately 220,000 gallons of borated water is required to provide adequate post-accident core cooling and containment spray to maintain calculated post-accident doses below the limits of 10 CFR 100[1]. The refueling water storage tank filled to elevation 50 feet represents in excess of 240,000 gallons. 34 4/1/77

A boron concentration of 3750 ppm is required to meet the requirements of postulated steam line break.[2] A maximum boron concentration of 4300 ppm ensures that the post-accident containment sump water is maintained at a pH between 7.0 and 7.5[3].

The refueling tank capacity of 240,000 gallons is based on refueling volume requirements.

Sustained temperatures below 32°F do not occur at San Onofre. At 32°F, boric acid is soluble up to approximately 4650 ppm boron. Therefore, no special provisions for temperature control to avoid either freezing or boron precipitation are necessary. 34 4/1/77

References:

(1) Enclosure 1 "Post-Accident Pressure Reanalysis, San Onofre Unit 1" to letter dated January 19, 1977 in Docket No. 50-206.

(2) "Steam Line Break Accident Reanalysis, San Onofre Nuclear Generating Station, Unit 1, October 1976" submitted by letter dated December 30, 1976 in Docket No. 50-206.

(3) Additional information, San Onofre, Unit 1 submitted by letter dated March 24, 1977 in Docket No. 50-206.

3.5 INSTRUMENTATION AND CONTROL

3.5.1 REACTOR TRIP SYSTEM INSTRUMENTATION

APPLICABILITY: As shown in Table 3.5.1-1.

OBJECTIVE: To delineate the conditions of the Plant instrumentation and safety circuits necessary to ensure reactor safety. 83 11/2/84

SPECIFICATION: As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.5.1-1 shall be OPERABLE.

ACTION: As shown in Table 3.5.1-1.

BASIS: During plant operations, the complete instrumentation systems will normally be in service.(1) Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits.(2) Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design.(1)(3) This Standard outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor control and protection system when any one or more of the channels is out of service. 117 12/13/88

References:

(1) Final Engineering Report and Safety Analysis, Section 6.

(2) Final Engineering Report and Safety Analysis, Section 6.2.

(3) NIS Safety Review Report, April 1988 117 12/13/88

TABLE 3.5.1-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTION UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	7
2. Power Range, Neutron Flux, Overpower Trip	4	2	3	1, 2	2#
3. Power Range, Neutron Flux, Dropped Rod Rod Stop	4	1**	4	1, 2	28#
4. Intermediate Range, Neutron Flux	2	1	2	1##, 2	3
5. Source Range, Neutron Flux					
A. Startup	2	1**	2	2##	4
B. Shutdown	2	1**	2	3*, 4*, 5*	7
C. Shutdown	2	0	1	3, 4, and 5	5
6. NIS Coincidentor Logic	2	1	2	1, 2	29
				3*, 4*, 5*	7
7. Pressurizer Variable Low Pressure	3	2	2	1###	6#
8. Pressurizer Fixed High Pressure	3	2	2	1, 2	6#
9. Pressurizer High Level	3	2	2	1	6#

TABLE 3.5.1-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTION UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. Reactor Coolant Flow					
A. Single Loop (Above 50% of Full Power)	1/loop	1/loop in any operating loop	1/loop in each operating loop	1	6#
B. Two Loops (Below 50% of Full Power)	1/loop	1/loop in two operating loops	1/loop in each operating loop	1####	6#
11. Steam/Feedwater Flow Mismatch	3	2	2	1,2	6#
12. Turbine Trip-Low Fluid Oil Pressure	3	2	2	1####	6#

TABLE 3.5.1-1 (Continued)

TABLE NOTATION

*	With the reactor trip system breakers in the closed position, the control rod drive system capable of rod withdrawal.	83 11/2/84
**	A "TRIP" will stop all rod withdrawal.	
#	The provisions of Specification 3.0.4 are not applicable.	
##	Below the Source Range High Voltage Cutoff Setpoint.	
###	Below the P-7 (At Power Reactor Trip Defeat) Setpoint.	
####	Above the P-7 (At Power Reactor Trip Defeat) Setpoint.	117 12/13/88

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are met: 83 11/2/84

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be returned to the untripped condition for up to 2 hours for surveillance testing of other channels per Specification 4.1. 117 12/13/88

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the Source Range High Voltage Cutoff Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the Source Range High Voltage Cutoff Setpoint. 83 11/2/84

b. Above the Source Range High Voltage Cutoff Setpoint but below 10 percent of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10 percent of RATED THERMAL POWER.

However, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE. 117 12/13/88

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes. 83 11/2/84

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.5.2 as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 8 hours. 83 11/2/84

ACTION 7 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 28 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirements, within one hour reduce THERMAL POWER such that T_{ave} is less than or equal to 551.5°F, and place the rod control system in manual mode. 117 12/13/88

ACTION 29 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirements, be in at least HOT STANDBY within 6 hours; however, one channel may be removed from service for up to 2 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

3.5.2 CONTROL ROD INSERTION LIMITS

APPLICABILITY: MODES 1 and 2 — 111 10/21/88

OBJECTIVE: This specification defines the insertion limits for the control rods in order to ensure (1) an acceptable core power distribution during power operation, (2) a limit on potential reactivity insertions for a hypothetical control rod ejection, and (3) core subcriticality after a reactor trip.

SPECIFICATION:

A. Except during low power physics tests or surveillance testing pursuant to Specification 4.1.1.G, the Shutdown Groups and Control Group 1 shall be fully withdrawn, and the position of Control Group 2 shall be at or above the 21-step uncertainty limit shown in Figure 3.5.2.1. — 111 10/21/88

B. The energy weighted average of the positions of Control Group 2 shall be at least 90% (i.e. > Step 288) withdrawn after the first 20% burnup of a core cycle. The average shall be computed at least twice every month and shall consist of all Control Group 2 positions during the core cycle. — 21 5/13/75

ACTION:

A. With the control groups inserted beyond the above insertion limits either:

1. Restore the control groups to within the limits within 2 hours, or

2. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the above figure, or

3. Be in at least HOT STANDBY within 6 hours. — 111 10/21/88

B. With a single dropped rod from a shutdown group or control group, the provisions of Action A are not applicable, and retrieval shall be performed without increasing THERMAL POWER beyond the THERMAL POWER level prior to dropping the rod. An evaluation of the effect of the dropped rod shall be made to establish permissible THERMAL POWER levels for continued operation. If retrieval is not successful within 3 hours from the time the rod was dropped, appropriate action, as determined from the evaluation, shall be taken. In no case shall operation longer than 3 hours be permitted if the dropped rod is worth more than 0.4% Δ k/k.

BASIS: During Startup and Power Operation, the shutdown groups and control group 1 are fully withdrawn and control of the reactor is maintained by control group 2. The control group insertion limits are set in consideration of maximum specific

power, shutdown capability, and the rod ejection accident. The considerations associated with each of these quantities are as follows:

1. The initial design maximum value of specific power is 15 kW/ft. The values of $F^N_{\Delta H}$ and F_Q total associated with this specific power are 1.75 and 3.23, respectively.

   A more restrictive limit on the design value of specific power, $F^N_{\Delta H}$ and F_Q is applied to operation in accordance with the current safety analysis including fuel densification and ECCS performance. The values of the specific power, $F^N_{\Delta H}$ and F_Q are 13.7 kW/ft, 1.57 and 2.89, respectively. At partial power, the $F^N_{\Delta H}$ maximum values (limits) increase according to the following equation, $F^N_{\Delta H}(P) = 1.57\,[1 + 0.2\,(1-P)]$, where P is the fraction of RATED THERMAL POWER. The control group insertion limits in conjunction with Specification B prevent exceeding these values even assuming the most adverse Xe distribution. | 60 6/8/81 | 88 11/23/84 | 111 10/21/88 | 60 6/8/81

2. The minimum shutdown capability required is 1.25% Δρ at BOL, 1.9% Δρ at EOL and defined linearly between these values for intermediate cycle lifetimes. The rod insertion limits ensure that the available SHUTDOWN MARGIN is greater than the above values. | 21 5/13/75 | 111 10/21/88

3. The worst case ejected rod accident (8) covering HFP-BOL, HZP-BOL, HFP-EOL shall satisfy the following accident safety criteria:

   a) Average fuel pellet enthalpy at the hot spot below 225 cal/gm for nonirradiated fuel and 220 cal/gm for irradiated fuel. | 54 5/29/80

   b) Fuel melting is limited to less than the innermost 10% of the fuel pellet at the hot spot.

   Low power physics tests are conducted approximately one to four times during the core cycle at or below 10% RATED THERMAL POWER. During such tests, rod configurations different from those specified in Figure 3.5.2.1 may be employed. | 111 10/21/88

   It is understood that other rod configurations may be used during physics tests. Such configurations are permissible based on the low probability of occurrence of steam line break or rod ejection during such rod configurations.

Operation of the reactor during cycle stretch out is conservative relative to the safety considerations of the control rod insertion limits, since the positioning of the rods during stretch out results in an increasing net available SHUTDOWN MARGIN.

Compliance with Specification B prevents unfavorable axial power distributions due to operation for long intervals at deep control rod insertions.

The presence of a dropped rod leads to abnormal power distribution in the core. The location of the rod and its worth in reactivity determines its effect on the temperatures of nearby fuel. Under certain conditions, continued operation could result in fuel damage, and it is the intent of ACTION B to avoid such damage. 111 10/21/88

References:

(1) Final Engineering Report and Safety Analysis, revised July 28, 1970.

(2) Amendment No. 18 to Docket No. 50-206.

(3) Amendment No. 22 to Docket No. 50-206.

(4) Amendment No. 23 to Docket No. 90-206.

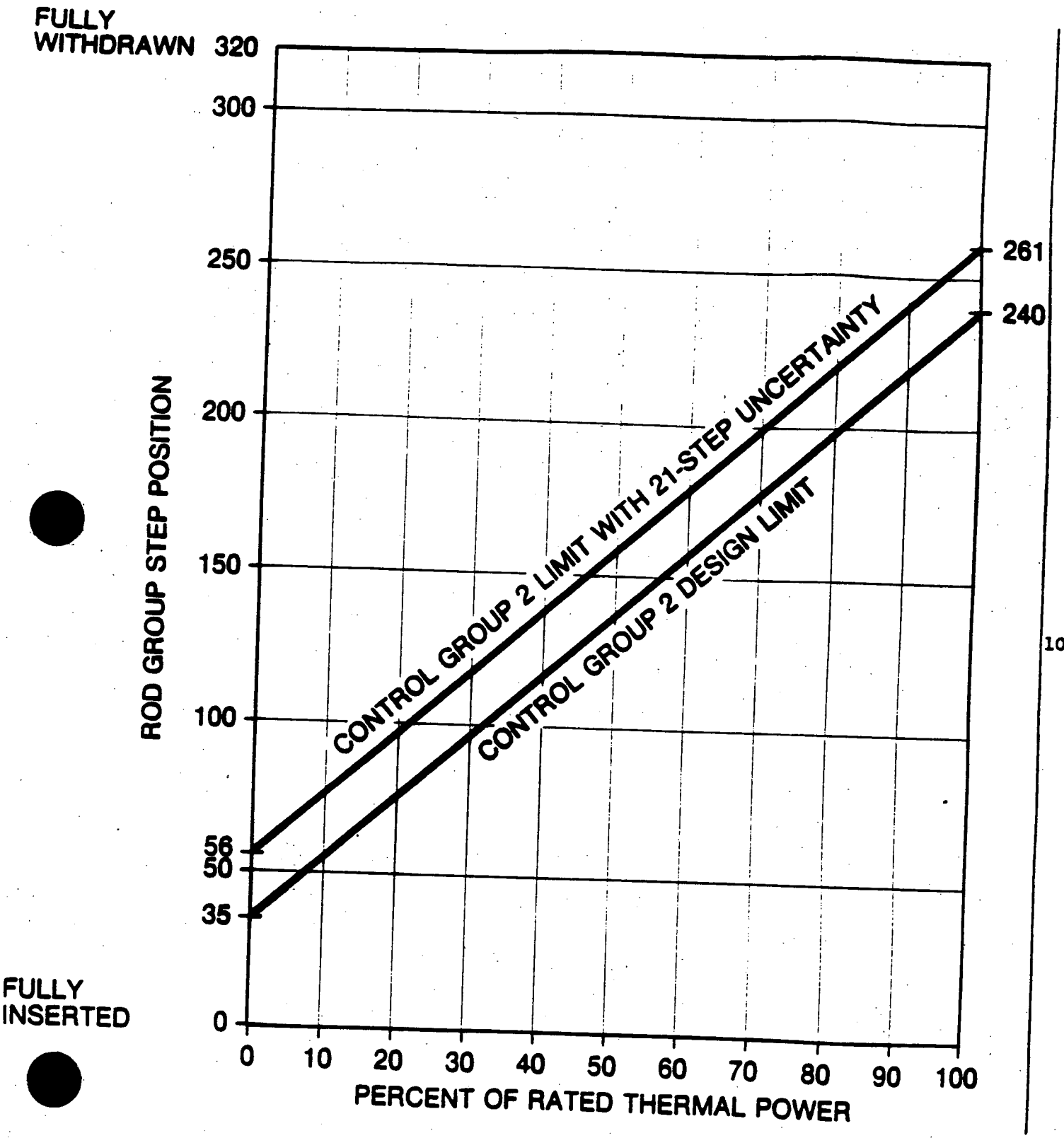
(5) Description and Safety Analysis, Proposed Change No. 7, dated October 22, 1971.

(6) Description and Safety Analysis Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1, Cycle 4, WCAP 8131, May, 1973.

(7) Description and Safety Analysis Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1, Cycle 5, January, 1975, Westinghouse Non-Proprietary Class 3.

(8) An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods, WCAP-7588, Revision 1-A, January, 1975.

CONTROL GROUP INSERTION LIMITS



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10/21/88

FIGURE 3.5.2.1

3.10 INCORE INSTRUMENTATION

112 10/28/88

APPLICABILITY: MODE 1 above 90% RATED THERMAL POWER

OBJECTIVE: To specify the type and frequency of incore measurements used to verify linear power density values.

SPECIFICATION:

a. A power distribution measurement shall be performed every 30 effective full power days (EFPD) and after attainment of equilibrium xenon upon return to power following a refueling shutdown. 112 10/28/88

b. The incore instrumentation system shall be used to accomplish the Correlation Verification of incore versus excore data for the axial offset monitoring system prior to exceeding 90% of RATED THERMAL POWER following each refueling and at least once per 180 effective full power days (EFPD) thereafter. Subsequent to the Correlation Verification and for the duration of each cycle, incore instrumentation shall be used to perform a Correlation Check of the axial offset monitoring system every 30 EFPD. 112 10/28/88

c. A core thermocouple map shall be taken every 30 EFPD and after attainment of equilibrium xenon upon return to power following a refueling shutdown. 112 10/28/88

ACTION:

A. If the correlation check, power distribution measurement or core thermocouple map described above cannot be made within the prescribed time, a maximum of 15 EFPD will be allowed for equipment correction. 112 10/28/88

B. In the event that Specification a, b and c cannot be met during the 15 EFPD allowed for corrective action, within one hour action shall be taken such that THERMAL POWER is restricted to less than or equal to 90% of RATED THERMAL POWER until these specifications can be met. 112 10/28/88

BASIS: The flux mapping system is used to measure the core power distribution and to correlate incore versus excore data for the axial offset system. Measurements made with the flux mapping system every 30 effective full power days and upon return to power following a refueling shutdown will monitor the core power distribution to confirm that the maximum linear power density remains below allowable values. The

axial offset system will monitor the axial core power distribution in a continuous manner. If the Correlation Verification or Correlation Check is not performed, the 90% of full thermal power restriction assures safe operation of the reactor. In addition, core thermocouples provide an independent means of measuring the balance of power among the core quadrants.

The flux mapping system and the thermocouple system are not integral parts of the Reactor Protection System. These systems are, rather, surveillance systems which may be required in the event of an abnormal condition such as a power tilt or a control rod misalignment. Since such a condition cannot be predicted, it is prudent to have the surveillance systems in an operable state. The 90% of full power restriction, used when these measurements cannot be taken as scheduled, is applied to minimize the probability of exceeding allowed peaking factors.

Operation for a 180 effective full power day period prior to reperforming the correlation verification is acceptable on the basis that the allowed incore axial offset limits are reduced by the amount in percent of incore axial offset that the monthly correlation check differs from the correlation. 112 10/28/88

3.11 CONTINUOUS POWER DISTRIBUTION MONITORING

112 10/28/88

APPLICABILITY: MODE 1 above 90% RATED THERMAL POWER

OBJECTIVE: To provide corrective action in the event that the axial offset monitoring system limits are approached.

SPECIFICATION: The incore axial offset limits shall not exceed the functional relationship defined by:

For positive offsets: $$IAO = \frac{2.89/P - 2.1225}{0.03021} - FCC$$

For negative offsets: $$IAO = \frac{2.89/P - 2.1181}{-.03068} + FCC$$

112 10/28/88

where

IAO = incore axial offset

P = fraction of RATED THERMAL POWER

FCC = The larger of 3.0 or the value in percent of incore axial offset by which the current correlation check differs from the incore-excore correlation.

112 10/28/88

ACTION:

A. With IAO exceeding the limit defined by the specification, within 1 hour action shall be taken to reduce THERMAL POWER until IAO is within specified limits or such that THERMAL POWER is restricted to less than 90% of RATED THERMAL POWER.

B. With one or both excore axial offset channel(s) inoperable, as an alternate, one OPERABLE NIS channel for each inoperable excore axial offset channel, shall be logged every two hours to determine IAO.

117 12/13/88

C. With no method for determining IAO available, within 1 hour action shall be taken such that THERMAL POWER is reduced to less than 90% of RATED THERMAL POWER until a method of determining axial offset is restored.

BASIS:

The percent full power axial offset limits are conservatively established considering the core design peaking factor, analytical determination of the relationship between core peaking factors and incore axial offset considering a wide range of maneuvers and core conditions, and actual measurements relating incore axial offset to the axial offset monitoring systems. The axial offset limit established from the incore versus excore data have been reduced by an amount equivalent to FCC to allow for burnup and time dependent differences between the periodic correlation verification and the monthly correlation check. Correcting the allowed incore axial offset limits by an amount equal to FCC maintains plant operation within the original safety analysis assumptions. Should a specific cycle analysis establish that the analytical determination of the relationship between core peaking factors and incore axial offset has changed in a manner warranting modification to the existing envelope of peaking factor (1,2), then a change to functional relationship of the specification shall be submitted to the Commission. The incore-excore data correlation is checked or verified periodically as delineated in Specification 3.10, INCORE INSTRUMENTATION. 112 10/28/88

Reducing power in cases when limits are approached or exceeded, will assure that design limits which were set in consideration of accident conditions are not exceeded. In the event that no method exists for determining IAO, actions are specified to reduce THERMAL POWER to 90% of RATED THERMAL POWER. However, if axial offset channel(s) are inoperable, hand calculational methods of determining IAO from OPERABLE NIS channels can be employed until OPERABILITY of the axial offset channel(s) is restored. 117 12/13/88

References:

(1) Supporting Information on Periodic Axial Offset Monitoring, San Onofre Nuclear Generating Station, Unit 1, September, 1973

(2) Supporting Information on the Continuous Axial Offset Monitoring System, San Onofre Nuclear Generating Station, Unit 1, July, 1974

(3) Description and Safety Analysis, Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1 Cycle 5, January, 1975, Westinghouse Non-Proprietary Class 3.

4.1.1 OPERATIONAL SAFETY ITEMS

Applicability: Applies to surveillance requirements for items directly related to Safety Standards and Limiting Conditions for Operation.

Objective: To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

Specification:

A. Reactor Trip System instrumentation shall be checked, tested, and calibrated as indicated in Table 4.1.1. — 117 12/13/88

B. Equipment and sampling tests shall be as specified in Table 4.1.2.

C. The specific activity and boron concentration of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.1.2., Item 1a. — 38 12/20/77

D. The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.1.2., Item 1b.

E. All control rods shall be determined to be above the rod insertion limits shown in Figure 3.5.2.1 by verifying that each analog detector indicates at least 21 steps above the rod insertion limits, to account for the instrument inaccuracies, at least once per shift during Startup conditions with K_{eff} equal to or greater than one.

F. The position of each rod shall be determined to be within the group demand limit and each rod position indicator shall be determined to be OPERABLE by verifying that the rod position indication system (Analog Detection System) and the step counter indication system (Digital Detection System) agree within 35 steps at least once per shift during Startup and Power Operation except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the rod position indication system (Analog Detection System) and the step counter indication system (Digital Detection System) at least once per 4 hours. — 61 6/11/81

G. During MODE 1 or 2 operation each rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days. — 83 11/2/84

H. Instrumentation shall be checked, tested, and calibrated as indicated in Table 4.1.3. — 117 12/13/88

TABLE 4.1.1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R	N.A.
2. Power Range, Neutron Flux	S	D (2,3) R (3,4)	M	N.A.	N.A.
3. Power Range, Neutron Flux, Dropped Rod Rod Stop	N.A.	N.A.	M	N.A.	N.A.
4. Intermediate Range, Neutron Flux	S	R (3,4)	S/U (1), M	N.A.	N.A.
5. Source Range, Neutron Flux	S	R (3)	S/U (1), M	N.A.	N.A.
6. NIS Coincidentor Logic	N.A.	N.A.	N.A.	N.A.	M (5)
7. Pressurizer Variable Low Pressure	S	R	M	N.A.	N.A.
8. Pressurizer Pressure	S	R	M	N.A.	N.A.
9. Pressurizer Level	S	R	M	N.A.	N.A.
10. Reactor Coolant Flow	S	R	Q	N.A.	N.A.
11. Steam/Feedwater Flow Mismatch	S	R	M	N.A.	N.A.
12. Turbine Trip-Low Fluid Oil Pressure	N.A.	N.A.	N.A.	S/U (1,6)	N.A.

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TABLE 4.1.1 (Continued)

TABLE NOTATION

(1) - If not performed in previous 31 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER. Adjust channel if absolute difference greater than 2 percent.

(3) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(4) - The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(5) - Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(6) - Setpoint verification is not applicable.

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TABLE 4.1.2
MINIMUM EQUIPMENT CHECK AND SAMPLING FREQUENCY

		Check	Frequency	
1a. Reactor Coolant Samples	1.	Gross Activity Determination	At least once per 72 hours. Required during Modes 1, 2, 3 and 4.	96 3/5/87
	2.	Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days. Required only during Mode 1.	
	3.	Spectroscopic for E(1) Determination	1 per 6 months(2) Required only during Mode 1.	74 12/6/84
	4.	Isotopic Analysis for Iodine Including I-131, I-133, and I-135.	a) Once per 4 hours,(3) whenever the specific activity exceeds 1.0 µCi/gram DOSE EQUIVALENT I-131 or 100/ E (1) µCi/gram.	
			b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	38 12/20/77
	5.	Boron concentration	Twice/Week	

(1) E is defined in Section 1.0. 117 12/13/88

(2) Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

(3) Until the specific activity of the reactor coolant system is restored within its limits.

TABLE 4.1.2 (continued)

		Check	Frequency	
1.b	Secondary Coolant Samples	1. Gross Activity Determination	At least once per 72 hours. Required only during Modes 1, 2, 3 and 4.	
		2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. Required only during Modes 1, 2, 3 and 4.	74 12/6/83
			b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. Required only during Modes 1, 2, 3, and 4.	

TABLE 4.1.2 (continued)

		Check		Frequency	
2.	Safety Injection Water Samples	a.	Boron Concentration	Monthly when the reactor is critical and prior to return of criticality when a period of subcriticality extends the test beyond 1 month	12 9/17/73
3.	Control Rod Drop	a.	Verify that all rods move from full out to full in, in less than 2.44 seconds	At each refueling shutdown	101 4/26/88
4.	(Deleted)				61 6/11/81
5.	Pressurizer Safety Valves	a.	Pressure Setpoint	At each refueling shutdown	
6.	Main Steam Safety Valves	a.	Pressure Setpoint	At each refueling shutdown	
7.	Main Steam Power Operated Relief Valves	a.	Test for Operability	At each refueling shutdown	
8.	Trisodium Phosphate Additive	a.	Check for system availability as delineated in Technical Specification 4.2	At each refueling shutdown	
9.	Hydrazine Tank Water Samples	a.	Hydrazine concentration	Once every six months when the reactor is critical and prior to return of criticality when a period of subcriticality extends the test interval beyond six months	34 4/1/77
10.	Transfer Switch No. 7	a.	Verify that the fuse block for breaker 8-1181 to MCC 1 is removed	Monthly, when the reactor is critical and prior to returning reactor to critical when period of subcriticality extended the test interval beyond one month	

TABLE 4.1.2 (continued)

		Check	Frequency	
11.	MOV-LCV-1100 C Transfer Switch	a. Verify that the fuse block for either breaker 8-1198 to MCC 1 or breaker 42-12A76 to MCC 2A is removed.	Same as Item 10 above	
12.	Emergency Siren Transfer Switch	a. Verify that the fuse block for either breaker 8-1145 to MCC 1 or breaker 8-1293A to MCC 2 is removed	Same as Item 10 above	34 4/1/77
13.	Communication Power Panel Transfer Switch	a. Verify that the fuse block for either breaker 8-1195 to MCC 1 or breaker 8-1293B to MCC 2 is removed	Same as Item 10 above	
14a.	Spent Fuel Pool Water Level	Verify water level per Technical Specification 3.8	a. Once every seven days when spent fuel is being stored in the pool.	
b.	Refueling Pool Water Level		b. Within two hours prior to start of and at least once per 24 hours thereafter during movement of fuel assemblies or RCC's.	43 9/25/78
15.	Reactor Coolant Loops/ Residual Heat Removal Loops	a. Per Technical Specifications 3.1.2.C and 3.1.2.D, in Mode 1 and Mode 2 and in Mode 3 with reactor trip breakers closed, verify that all required reactor coolant loops are in operation and circulating reactor coolant. b. Per Technical Specification 3.1.2.E, in Mode 3 with the reactor trip breakers open, verify	a. Once per 12 hours	104 6/9/8

TABLE 4.1.2 (Continued)

Check	Frequency
1. At least two required reactor coolant pumps are operable with correct breaker alignments and indicated power availability.	1. Once per 7 days
2. The steam generators associated with the two required reactor coolant pumps are operable with secondary side water level $\geq$ 256 inches of narrow range on cold calibrated scale.	2. Once per 12 hours
3. At least one reactor coolant loop is in operation and circulating reactor coolant.	3. Once per 12 hours
c. Per Technical Specification 3.1.2.F, in Mode 4 verify	
1. At least two required (RC or RHR) pumps are operable with correct breaker alignments and indicated power availability.	1. Once per 7 days
2. The required steam generators are operable with secondary side water level $\geq$ 256 inches of narrow range on cold calibrated scale.	2. Once per 12 hours
3. At least one reactor coolant loop/RHR train is in operation and circulating reactor coolant.	3. Once per 12 hours
d. Per Technical Specifications 3.1.2.G and 3.1.2.H, in Mode 5 verify, as applicable:	

80
10/4/84

TABLE 4.1.2 (Continued)

	Check	Frequency	
	1. At least one RHR train is in operation and circulating reactor coolant.	1. Once per 12 hours	
	2. When required, one additional RHR train is operable with correct pump breaker alignments and indicated power availability.	2. Once per 7 days	
	3. When required, the secondary side water level of at least two steam generators is $\geq$ 256 inches of narrow range on cold calibrated scale.	3. Once per 12 hours	
e.	Per Technical Specification 3.8.A.3, in Mode 6, with water level in refueling pool greater than elevation 40 feet 3 inches, verify that at least one method of decay heat removal is in operation and circulating reactor coolant at a flow rate of at least 400 gpm.	e. Once per 12 hours	80 10/4/84
f.	Per Technical Specification 3.8.A.4, in Mode 6, with water level in refueling pool less than elevation 40 feet 3 inches, verify		
	1. At least one decay heat removal method is in operation and circulating reactor coolant.	1. Once per 12 hours	
	2. One additional decay heat removal method is operable with correct pump breaker alignments and indicated power availability.	2. Once per 7 days	

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATIONS

2.1 REACTOR CORE - Limiting Combination of Power, Pressure, and Temperature

APPLICABILITY: Applies to reactor power, system pressure, coolant temperature, and flow during operation of the plant.

OBJECTIVE: To maintain the integrity of the reactor coolant system and to prevent the release of excessive amounts of fission product activity to the coolant.

SPECIFICATION: Safety Limits

(1) The reactor coolant system pressure shall not exceed 2735 psig with fuel assemblies in the reactor.

(2) The combination of reactor power and coolant temperature shall not exceed the locus of points established for the RCS pressure in Figure 2.1.1. If the actual power and temperature is above the locus of points for the appropriate RCS pressure, the safety limit is exceeded.

Maximum Safety System Settings

The maximum safety system trip settings shall be as stated in Table 2.1.

BASIS: Safety Limits

1. Reactor Coolant System Pressure

The Reactor Coolant System serves as a barrier which prevents release of radionuclides contained in the reactor coolant to the containment atmosphere. In addition, the failure of components of the Reactor Coolant System could result in damage to the fuel and pressurization of the containment. A safety limit of 2735 psig (110% of design pressure) has been established which represents the maximum transient pressure allowable in the Reactor Coolant System under the ASME Code, Section VIII.

2. Plant Operating Transients

In order to prevent any significant amount of fission products from being released from the fuel to the reactor coolant, it is necessary to prevent clad overheating both during normal operation and while undergoing system transients. Clad overheating and potential failure could occur if the heat transfer mechanism at the clad surface departs from nucleate boiling. System parameters which affect this departure from nucleate boiling (DNB) have been correlated with experimental data to provide a means

of determining the probability of DNB occurrence. The ratio of the heat flux at which DNB is expected to occur for a given set of conditions to the actual heat flux experienced at a point is the DNB ratio and reflects the probability that DNB will actually occur.

It has been determined that under the most unfavorable conditions of power distribution expected during core lifetime and if a DNB ratio of 1.44 should exist, not more than 7 out of the total of 28,260 fuel rods would be expected to experience DNB. These conditions correspond to a reactor power of 125% of rated power. Thus, with the expected power distribution and peaking factors, no significant release of fission products to the reactor coolant system should occur at DNB ratios greater than 1.30.(1) The DNB ratio, although fundamental, is not an observable variable. For this reason, limits have been placed on reactor coolant temperature, flow, pressure, and power level, these being the observable process variables related to determination of the DNB ratio. The curves presented in Figure 2.1.1 represent loci of conditions at which a minimum DNB ratio of 1.30 or greater would occur. (1)(2)(3)

Maximum Safety System Settings

1. Pressurizer High Level and High Pressure

In the event of loss of load, the temperature and pressure of the Reactor Coolant System would increase since there would be a large and rapid reduction in the heat extracted from the Reactor Coolant System through the steam generators. The maximum settings of the pressurizer high level trip and the pressurizer high pressure trip are established to maintain the DNB ratio above 1.30 and to prevent the loss of the cushioning effect of the steam volume in the pressurizer (resulting in a solid hydraulic system) during a loss-of-load transient.(3)(4)

In the event that steam/feedflow mismatch trip cannot be credited due to single failure considerations, the pressurizer high level trip is provided. In order to meet acceptance criteria for the Loss of Main Feedwater and Feedline Break transients, the pressurizer high level trip must be set at 20.8 ft. (50%) or less.

2. Variable Low Pressure Loss of Flow and Nuclear Overpower Trips

These settings are established to accommodate the most severe transients upon which the design is based, e.g., loss of coolant flow, rod withdrawal at power, control rod

ejection, inadvertent boron dilution and large load increase without exceeding the safety limits. The settings have been derived in consideration of instrument errors and response times of all necessary equipment. Thus, these settings should prevent the release of any significant quantities of fission products to the coolant as a result of transients.(3)(4)(5)(7)

In order to prevent significant fuel damage in the event of increased peaking factors due to an asymmetric power distribution in the core, the nuclear overpower trip setting on all channels is reduced by one percent for each percent that the asymmetry in power distribution exceeds 5%. This provision should maintain the DNB ratio above a value of 1.30 throughout design transients mentioned above.

The response of the plant to a reduction in coolant flow while the reactor is at substantial power is a corresponding increase in reactor coolant temperature. If the increase in temperature is large enough, DNB could occur, following loss of flow.

The low flow signal is set high enough to actuate a trip in time to prevent excessively high temperatures and low enough to reflect that a loss of flow conditions exists. Since coolant loop flow is either full on or full off, any loss of flow would mean a reduction of the initial flow (100%) to zero.(3)(6)

3. Reactor Coolant Pump Breaker Open

The Reactor Coolant Pump (RCP) Breaker Open reactor trip provides a redundant trip to the low flow trip. The overcurrent trip of the RCP breakers protects the core following a locked rotor and the undercurrent trip of the RCP breakers protects the core following a sheared shaft. The trip settings are selected to meet the analysis assumptions that rods begin to drop 6.1 seconds after the initiating event. The Reactor Protection System Permissives change the trip on RCP breaker open to 2/3 loops instead of 1/3 loops at power levels below 50%.

References:

(1) Amendment No. 10 to the Final Engineering Report and Safety Analysis, Section 4, Question 3

(2) Final Engineering Report and Safety Analysis, Paragraph 3.3

(3) Final Engineering Report and Safety Analysis, Paragraph 6.2

(4) Final Engineering Report and Safety Analysis, Paragraph 10.6

(5) Final Engineering Report and Safety Analysis, Paragraph 9.2

(6) Final Engineering Report and Safety Analysis, Paragraph 10.2

(7) NIS Safety Review Report, April 1988

(8) Reload Safety Evaluation, Cycle 10, Revision 1, March 1989, by Westinghouse, editor J. Skaritka

TABLE 2.1

MAXIMUM SAFETY SYSTEM SETTINGS

	Three Reactor Coolant Pumps Operating
*1. Pressurizer High Level	≤ 20.8 ft. above bottom of pressurizer when steam/feedflow mismatch trip is not credited, or ≤ 27.3 ft. above bottom of pressurizer when steam/feedflow mismatch trip is credited
2. Pressurizer Pressure: High	≤ 2220 psig
3. Nuclear Overpower	
a. High Setting**	≤ 109% of indicated full power
b. Low Setting	≤ 25% of indicated full power
***4. Variable Low Pressure	≥ 26.15 (0.894 ΔT+T avg.) - 14341
***5. Coolant Flow	≥ 85% of indicated full loop flow
6. Reactor Coolant Pump Breaker Open	
****a. Overcurrent	≤ 2900 amps at 4160 volts
****b. Undercurrent	≥ 110 amps at 4160 volts

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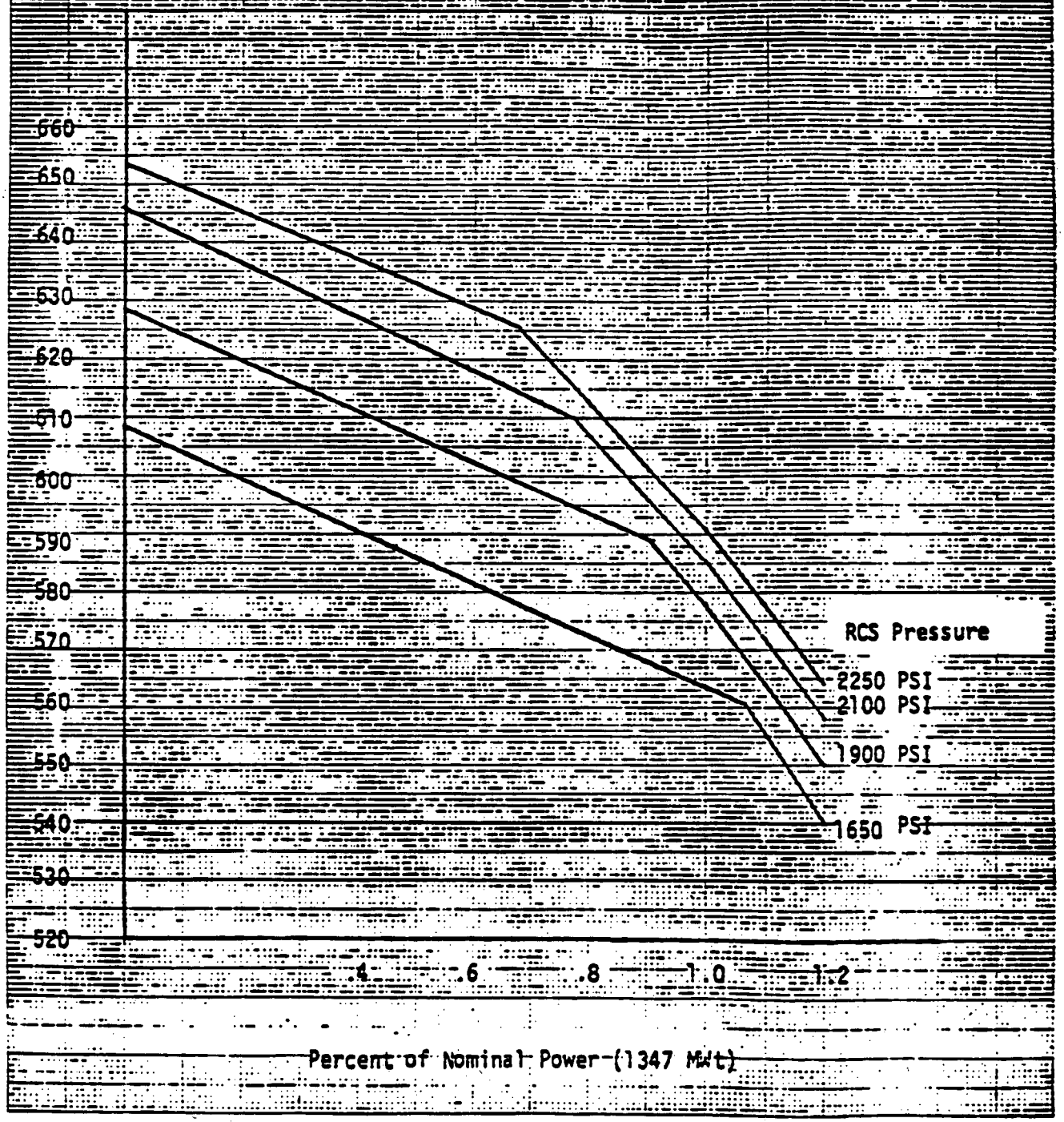
* Credit can be taken for the steam/feedflow mismatch trip when this system is modified such that a single failure will not prevent the system from performing its safety function.

** The nuclear overpower trip is based upon a symmetrical power distribution. If an asymmetric power distribution greater than 5% should occur, the nuclear overpower trip on all channels shall be reduced one percent for each percent above 5%.

*** May be bypassed at power levels below 10% of full power.

****May be bypassed at power levels below 50% of full power.

Figure 2.1.1
Safety Limits
Temperature, Power, Pressure
RCS Flow ≥ 195,000 GPM



60
6/8/81

Revised: 6/23/81

3.3.3 MINIMUM BORON CONCENTRATION IN THE REFUELING WATER STORAGE TANK (RWST) AND SAFETY INJECTION (SI) LINES AND MINIMUM RWST WATER VOLUME

APPLICABILITY: MODES 1, 2, 3 and 4; or as described in Specification 3.2.

OBJECTIVE: To ensure immediate availability of borated water from the RWST for safety injection, containment spray or emergency boration.

SPECIFICATION:

a. The RWST shall be OPERABLE with a level of at least plant elevation 50 feet of water having a boron concentration of not less than 3750 ppm and not greater than 4300 ppm.

b. The safety injection (SI) lines from the RWST to MOV 850 A, B, and C, with the exception of lines common to the feedwater system, shall be OPERABLE with a boron concentration of not less than 1500 ppm and not greater than 4300 ppm.

ACTION:

A. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

B. With one or both SI lines inoperable due to boron concentration of less than 1500 ppm, restore the SI lines to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.

BASIS:

The refueling water storage tank serves three purposes; namely:

(1) As a reservoir of borated water for accident mitigation purposes,

(2) As a reservoir of borated water for flooding the refueling cavity during refueling.

(3) As a deluge for fires in containment.

Approximately 220,000 gallons of borated water is required to provide adequate post-accident core cooling and containment spray to maintain calculated post-accident doses below the limits of 10 CFR 100[1]. The refueling water storage tank filled to a plant elevation 50 feet represents in excess of 240,000 gallons.

A boron concentration of 3750 ppm in the RWST and 1500 ppm in the SI lines is required to meet the requirements of a postulated steam line break.[2][4] A maximum boron concentration of 4300 ppm ensures that the post-accident containment sump water is maintained at a pH between 7.0 and 7.5[3].

The refueling tank capacity of 240,000 gallons is based on refueling volume requirements and includes an allowance for water not usable because of tank discharge line location.

Sustained temperatures below 32°F do not occur at San Onofre. At 32°F, boric acid is soluble up to approximately 4650 ppm boron. Therefore, no special provisions for temperature control to avoid either freezing or boron precipitation are necessary.

References:

(1) Enclosure 1 "Post-Accident Pressure Reanalysis, San Onofre Unit 1" to letter dated January 19, 1987 in Docket No. 50-206

(2) "Main Steamline Break Analysis, San Onofre Nuclear Generating Station, Unit 1, August 1988"

(3) Additional information, San Onofre, Unit 1 submitted by letter dated March 24, 1987 in Docket No. 50-206

(4) Reload Safety Evaluation, San Onofre Nuclear Generating Station, Unit 1, Cycle 10, edited by J. Skaritka, Revision 1, Westinghouse, March, 1989

3.5 INSTRUMENTATION AND CONTROL

3.5.1 REACTOR TRIP SYSTEM INSTRUMENTATION

APPLICABILITY: As shown in Table 3.5.1-1.

OBJECTIVE: To delineate the conditions of the Plant instrumentation and safety circuits necessary to ensure reactor safety.

SPECIFICATION: As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.5.1-1 shall be OPERABLE.

ACTION: As shown in Table 3.5.1-1.

BASIS: During plant operations, the complete instrumentation systems will normally be in service.(1) Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits.(2) Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design.(1)(3) This Standard outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor control and protection system when any one or more of the channels is out of service.

References:

(1) Final Engineering Report and Safety Analysis, Section 6.

(2) Final Engineering Report and Safety Analysis, Section 6.2.

(3) NIS Safety Review Report, April 1988

TABLE 3.5.1-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTION UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	7
2. Power Range, Neutron Flux, Overpower Trip	4	2	3	1, 2	2#
3. Power Range, Neutron Flux, Dropped Rod Rod Stop	4	1**	4	1, 2	28#
4. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
5. Source Range, Neutron Flux					
A. Startup	2	1**	2	2##	4
B. Shutdown	2	1**	2	3*, 4*, 5*	7
C. Shutdown	2	0	1	3, 4, and 5	5
6. NIS Coincidentor Logic	2	1	2	1, 2	29
				3*, 4*, 5*	7
7. Pressurizer Variable Low Pressure	3	2	2	1####	6#
8. Pressurizer Fixed High Pressure	3	2	2	1, 2	6#
9. Pressurizer High Level	3	2	2	1	6#

TABLE 3.5.1-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTION UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. Reactor Coolant Flow					
A. Single Loop (Above 50% of Full Power)	1/loop	1/loop in any operating loop	1/loop in each operating loop	1	6#
B. Two Loops (Below 50% of Full Power)	1/loop	1/loop in two operating loops	1/loop in each operating loop	1####	6#
11. Steam/Feedwater Flow Mismatch	3	2	2	1,2	6#
12. Turbine Trip-Low Fluid Oil Pressure	3	2	2	1####	6#
13. Reactor Coolant Pump Breaker Position (Above 50% of Full Power)	1/loop	1/loop	1/loop	1	6#

TABLE 3.5.1-1 (Continued)

TABLE NOTATION

*	With the reactor trip system breakers in the closed position, the control rod drive system capable of rod withdrawal.
**	A "TRIP" will stop all rod withdrawal.
#	The provisions of Specification 3.0.4 are not applicable.
##	Below the Source Range High Voltage Cutoff Setpoint.
###	Below the P-7 (At Power Reactor Trip Defeat) Setpoint.
####	Above the P-7 (At Power Reactor Trip Defeat) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are met:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be returned to the untripped condition for up to 2 hours for surveillance testing of other channels per Specification 4.1.

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the Source Range High Voltage Cutoff Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the Source Range High Voltage Cutoff Setpoint.

b. Above the Source Range High Voltage Cutoff Setpoint but below 10 percent of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10 percent of RATED THERMAL POWER.

However, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.5.2 as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 8 hours.

ACTION 7 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 28 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirements, within one hour reduce THERMAL POWER such that T_{ave} is less than or equal to 551.5°F, and place the rod control system in manual mode.

ACTION 29 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirements, be in at least HOT STANDBY within 6 hours; however, one channel may be removed from service for up to 2 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

3.5.2 CONTROL ROD INSERTION LIMITS

APPLICABILITY: MODES 1 and 2

OBJECTIVE: This specification defines the insertion limits for the control rods in order to ensure (1) an acceptable core power distribution during power operation, (2) a limit on potential reactivity insertions for a hypothetical control rod ejection, and (3) core subcriticality after a reactor trip.

SPECIFICATION:

A. Except during low power physics tests or surveillance testing pursuant to Specification 4.1.1.G, the Shutdown Groups and Control Group 1 shall be fully withdrawn, and the position of Control Group 2 shall be at or above the 21-step uncertainty limit shown in Figure 3.5.2.1.

B. The energy weighted average of the positions of Control Group 2 shall be at least 90% (i.e. > Step 288) withdrawn after the first 20% burnup of a core cycle. The average shall be computed at least twice every month and shall consist of all Control Group 2 positions during the core cycle.

ACTION:

A. With the control groups inserted beyond the above insertion limits either:

1. Restore the control groups to within the limits within 2 hours, or

2. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the above figure, or

3. Be in at least HOT STANDBY within 6 hours.

B. With a single dropped rod from a Shutdown Group or Control Group, the provisions of Action A are not applicable, and retrieval shall be performed without increasing THERMAL POWER beyond the THERMAL POWER level prior to dropping the rod. An evaluation of the effect of the dropped rod shall be made to establish permissible THERMAL POWER levels for continued operation. If retrieval is not successful within 3 hours from the time the rod was dropped, appropriate action, as determined from the evaluation, shall be taken. In no case shall operation longer than 3 hours be permitted if the dropped rod is worth more than 0.4% Δ k/k.

BASIS: During Startup and Power Operation, the Shutdown Groups and Control Group 1 are fully withdrawn and control of the reactor is maintained by Control Group 2. The Control Group insertion limits are set in consideration of maximum specific

power, shutdown capability, and the rod ejection accident. The considerations associated with each of these quantities are as follows:

1. The initial design maximum value of specific power is 15 kW/ft. The values of $F^N_{\Delta H}$ and F_Q total associated with this specific power are 1.75 and 3.23, respectively.

   A more restrictive limit on the design value of specific power, $F^N_{\Delta H}$ and F_Q is applied to operation in accordance with the current safety analysis including fuel densification and ECCS performance. The values of the specific power, $F^N_{\Delta H}$ and F_Q are 13.2 kW/ft, 1.57 and 2.78, respectively (8). At partial power, the $F^N_{\Delta H}$ maximum values (limits) increase according to the following equation,
   $F^N_{\Delta H}(P) = 1.57\ [1 + 0.2\ (1-P)]$, where P is the fraction of RATED THERMAL POWER. The Control Group insertion limits in conjunction with Specification B prevent exceeding these values even assuming the most adverse Xe distribution.

2. The minimum shutdown capability required is 1.25% $\Delta\rho$ at BOL, 1.9% $\Delta\rho$ at EOL and defined linearly between these values for intermediate cycle lifetimes. The rod insertion limits ensure that the available SHUTDOWN MARGIN is greater than the above values.

3. The worst case ejected rod accident (9) covering HFP-BOL, HZP-BOL, HFP-EOL shall satisfy the following accident safety criteria:

   a) Average fuel pellet enthalpy at the hot spot below 225 cal/gm for nonirradiated fuel and 220 cal/gm for irradiated fuel.

   b) Fuel melting is limited to less than the innermost 10% of the fuel pellet at the hot spot.

   Low power physics tests are conducted approximately one to four times during the core cycle at or below 10% RATED THERMAL POWER. During such tests, rod configurations different from those specified in Figure 3.5.2.1 may be employed.

   It is understood that other rod configurations may be used during physics tests. Such configurations are permissible based on the low probability of occurrence of steam line break or rod ejection during such rod configurations.

Operation of the reactor during cycle stretch out is conservative relative to the safety considerations of the control rod insertion limits, since the positioning of the rods during stretch out results in an increasing net available SHUTDOWN MARGIN.

Compliance with Specification B prevents unfavorable axial power distributions due to operation for long intervals at deep control rod insertions.

The presence of a dropped rod leads to abnormal power distribution in the core. The location of the rod and its worth in reactivity determines its effect on the temperatures of nearby fuel. Under certain conditions, continued operation could result in fuel damage, and it is the intent of ACTION B to avoid such damage.

References:

(1) Final Engineering Report and Safety Analysis, revised July 28, 1970.

(2) Amendment No. 18 to Docket No. 50-206.

(3) Amendment No. 22 to Docket No. 50-206.

(4) Amendment No. 23 to Docket No. 90-206.

(5) Description and Safety Analysis, Proposed Change No. 7, dated October 22, 1971.

(6) Description and Safety Analysis Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1, Cycle 4, WCAP 8131, May, 1973.

(7) Description and Safety Analysis Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1, Cycle 5, January, 1975, Westinghouse Non-Proprietary Class 3.

(8) Reload Safety Evaluation, San Onofre Nuclear Generating Station, Unit 1, Cycle 10, edited by J. Skaritka, Revision 1, Westinghouse, March, 1989

(9) An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods, WCAP-7588, Revision 1-A, January, 1975.

CONTROL GROUP INSERTION LIMITS

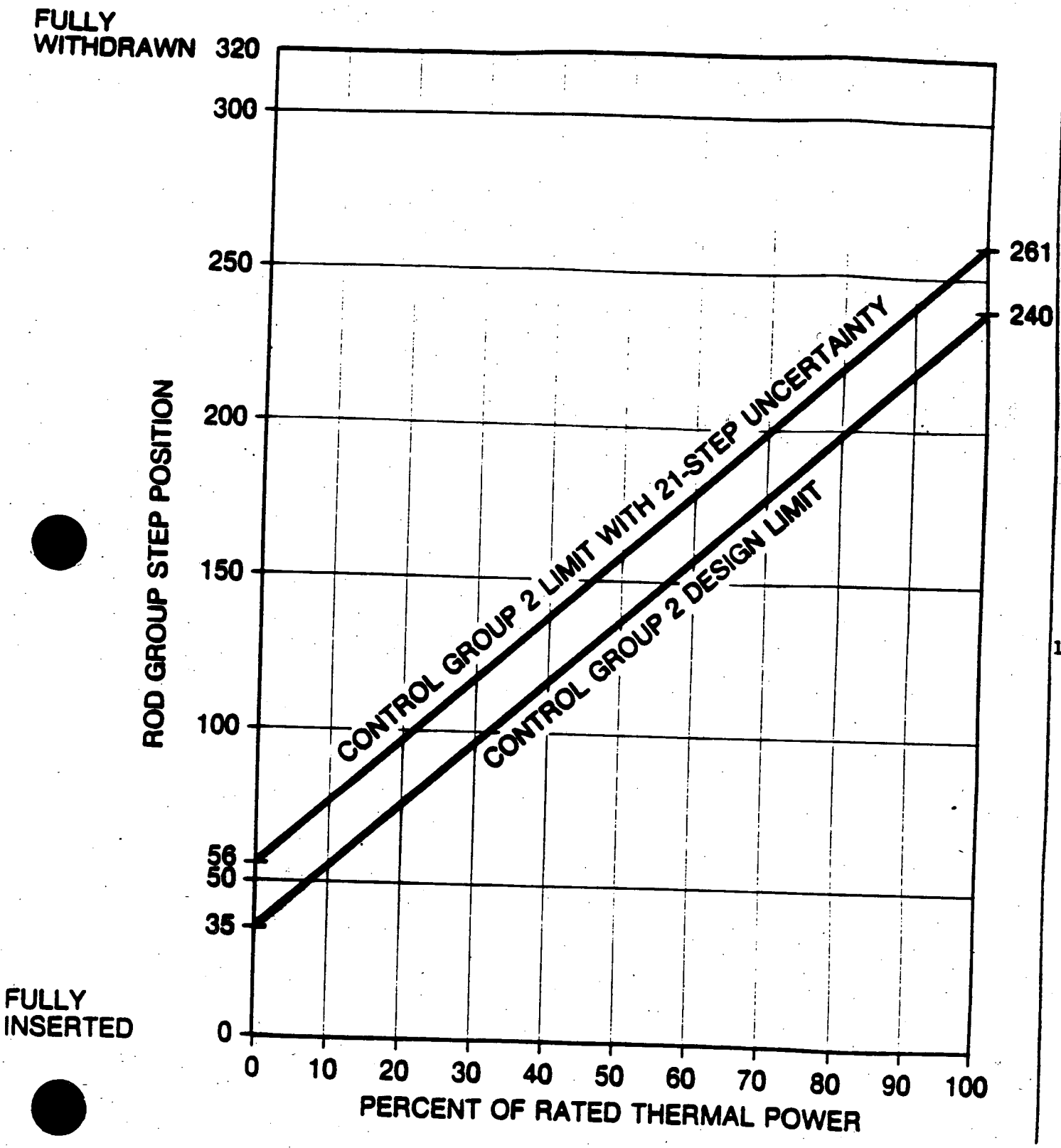


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FIGURE 3.5.2.1

3.10 INCORE INSTRUMENTATION

APPLICABILITY: MODE 1

OBJECTIVE: To specify the type and frequency of incore measurements used to verify linear power density values.

SPECIFICATION:

a. A power distribution measurement shall be performed every 30 Effective Full Power Days (EFPDs) and after attainment of equilibrium xenon upon return to power following a refueling shutdown.

b. The incore instrumentation system shall be used to accomplish the Correlation Verification of incore versus excore data for the axial offset monitoring system prior to exceeding 90% of RATED THERMAL POWER following each refueling and at least once per 180 EFPDs thereafter. Subsequent to the Correlation Verification and for the duration of each cycle, incore instrumentation shall be used to perform a Correlation Check of the axial offset monitoring system every 30 EFPDs.

c. A core thermocouple map shall be taken every 30 EFPDs and after attainment of equilibrium xenon upon return to power following a refueling shutdown.

ACTION:

A. If the correlation check, power distribution measurement or core thermocouple map described above cannot be made within the prescribed time, a maximum of 15 EFPDs will be allowed for equipment correction.

B. In the event that Specification a, b and c cannot be met during the 15 EFPDs allowed for corrective action, be in MODE 2 within 6 hours.

BASIS: The flux mapping system is used to measure the core power distribution and to correlate incore versus excore data for the axial offset system. Measurements made with the flux mapping system every 30 EFPDs and upon return to power following a refueling shutdown will monitor the core power distribution to confirm that the maximum linear power density remains below allowable values. The axial offset system will monitor the axial core power distribution in a continuous manner. In addition, core thermocouples provide an independent means of measuring the balance of power among the core quadrants.

The flux mapping system and the thermocouple system are not integral parts of the Reactor Protection System. These systems are, rather, surveillance systems which may be required in the event of an abnormal condition such as a power tilt or a control rod misalignment. Since such a condition cannot be predicted, it is prudent to have the surveillance systems OPERABLE. If the measurements cannot be taken as specified, the plant will be placed in MODE 2 within 6 hours as specified by the actions.

Operation for a 180 EFPD period prior to reperforming the correlation verification is acceptable on the basis that the allowed incore axial offset limits are reduced by the amount in percent of incore axial offset that the monthly correlation check differs from the correlation.

3.11 CONTINUOUS POWER DISTRIBUTION MONITORING

APPLICABILITY: MODE 1

OBJECTIVE: To provide corrective action in the event that the axial offset monitoring system limits are approached.

SPECIFICATION: The incore axial offset limits shall not exceed the functional relationship defined by:

For positive offsets: $$IAO = \frac{2.78/P - 2.10}{0.033} - FCC$$

For negative offsets: $$IAO = \frac{2.78/P - 2.10}{-0.033} + FCC$$

where

IAO = Incore Axial Offset

P = fraction of RATED THERMAL POWER

FCC = The larger of 3.0 or the value in percent of IAO by which the current correlation check differs from the incore-excore correlation.

ACTION:

A. With IAO exceeding the limit defined by the specification, within 1 hour action shall be taken to reduce THERMAL POWER until IAO is within specified limits.

B. With one or both excore axial offset channel(s) inoperable, as an alternate, one OPERABLE NIS channel for each inoperable excore axial offset channel, shall be logged every two hours to determine IAO.

C. With no method for determining IAO available, be in MODE 2 within 6 hours.

BASIS:

The percent full power axial offset limits are conservatively established considering the core design peaking factor, analytical determination of the relationship between core peaking factors and IAO considering a wide range of maneuvers and core conditions, and actual measurements relating IAO to the axial offset monitoring systems (1). The axial offset limit established from the incore versus excore data have been reduced by an amount equivalent to FCC to allow for burnup and time dependent differences between the periodic correlation verification and the monthly correlation check. Correcting the allowed IAO limits by an amount equal to FCC maintains plant operation within the original safety analysis assumptions. Should a specific cycle analysis establish that the analytical determination of the relationship between core peaking factors and IAO has changed in a manner warranting modification to the existing envelope of peaking factor (1,2), then a change to functional relationship of the specification shall be submitted to the Commission. The incore-excore data correlation is checked or verified periodically as delineated in Specification 3.10, INCORE INSTRUMENTATION.

Reducing power until IAO is within the specified limits in cases when limits are exceeded, will assure that design limits which were set in consideration of accident conditions are not exceeded. In the event that no method exists for determining IAO, actions are specified to place the plant in MODE 2 within 6 hours. However, if axial offset channel(s) are inoperable, hand calculational methods of determining IAO from OPERABLE NIS channels can be employed until OPERABILITY of the axial offset channel(s) is restored.

References:

(1) Reload Safety Evaluation, San Onofre Nuclear Generating Station, Unit 1, Cycle 10, edited by J. Skaritka, Revision 1, Westinghouse, March, 1989

(2) Supporting Information on Periodic Axial Offset Monitoring, San Onofre Nuclear Generating Station, Unit 1, September, 1973

(3) Supporting Information on the Continuous Axial Offset Monitoring System, San Onofre Nuclear Generating Station, Unit 1, July, 1974

(4) Description and Safety Analysis, Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1 Cycle 5, January, 1975, Westinghouse Non-Proprietary Class 3.

4.1.1 OPERATIONAL SAFETY ITEMS

Applicability: Applies to surveillance requirements for items directly related to Safety Standards and Limiting Conditions for Operation.

Objective: To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

Specification:

A. Reactor Trip System instrumentation shall be checked, tested, and calibrated as indicated in Table 4.1.1.

B. Equipment and sampling tests shall be as specified in Table 4.1.2.

C. The specific activity and boron concentration of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.1.2., Item 1a.

D. The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.1.2., Item 1b.

E. All control rods shall be determined to be above the rod insertion limits shown in Figure 3.5.2.1 by verifying that each analog detector indicates at least 21 steps above the rod insertion limits, to account for the instrument inaccuracies, at least once per shift during Startup conditions with K_{eff} equal to or greater than one.

F. The position of each rod shall be determined to be within the group demand limit and each rod position indicator shall be determined to be OPERABLE by verifying that the rod position indication system (Analog Detection System) and the step counter indication system (Digital Detection System) agree within 35 steps at least once per shift during Startup and Power Operation except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the rod position indication system (Analog Detection System) and the step counter indication system (Digital Detection System) at least once per 4 hours.

G. During MODE 1 or 2 operation each rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

H. Instrumentation shall be checked, tested, and calibrated as indicated in Table 4.1.3.

TABLE 4.1.1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R	N.A.
2. Power Range, Neutron Flux	S	D (2,3) R (3,4)	M	N.A.	N.A.
3. Power Range, Neutron Flux, Dropped Rod Rod Stop	N.A.	N.A.	M	N.A.	N.A.
4. Intermediate Range, Neutron Flux	S	R (3,4)	S/U (1), M	N.A.	N.A.
5. Source Range, Neutron Flux	S	R (3)	S/U (1), M	N.A.	N.A.
6. NIS Coincidentor Logic	N.A.	N.A.	N.A.	N.A.	M (5)
7. Pressurizer Variable Low Pressure	S	R	M	N.A.	N.A.
8. Pressurizer Pressure	S	R	M	N.A.	N.A.
9. Pressurizer Level	S	R	M	N.A.	N.A.
10. Reactor Coolant Flow	S	R	Q	N.A.	N.A.
11. Steam/Feedwater Flow Mismatch	S	R	M	N.A.	N.A.
12. Turbine Trip-Low Fluid Oil Pressure	N.A.	N.A.	N.A.	S/U (1,6)	N.A.
13. Reactor Coolant Pump Breaker Position	S	R	R	N.A.	N.A.

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TABLE 4.1.1 (Continued)

TABLE NOTATION

(1) - If not performed in previous 31 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER. Adjust channel if absolute difference greater than 2 percent.

(3) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(4) - The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(5) - Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(6) - Setpoint verification is not applicable.

TABLE 4.1.2
MINIMUM EQUIPMENT CHECK AND SAMPLING FREQUENCY

		Check	Frequency
1a. Reactor Coolant Samples	1.	Gross Activity Determination	At least once per 72 hours. Required during Modes 1, 2, 3 and 4.
	2.	Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days. Required only during Mode 1.
	3.	Spectroscopic for E(1) Determination	1 per 6 months(2) Required only during Mode 1.
	4.	Isotopic Analysis for Iodine Including I-131, I-133, and I-135.	a) Once per 4 hours,(3) whenever the specific activity exceeds 1.0 μCi/gram DOSE EQUIVALENT I-131 or 100/ E (1) μCi/gram. b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.
	5.	Boron concentration	Twice/Week

(1) E is defined in Section 1.0.

(2) Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

(3) Until the specific activity of the reactor coolant system is restored within its limits.

TABLE 4.1.2 (continued)

		Check	Frequency
1.b	Secondary Coolant Samples	1. Gross Activity Determination	At least once per 72 hours. Required only during Modes 1, 2, 3 and 4.
		2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. Required only during Modes 1, 2, 3 and 4. b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. Required only during Modes 1, 2, 3, and 4.

TABLE 4.1.2 (continued)

			Check	Frequency
2.	Safety Injection Line and RWST Water Samples	a.	Boron Concentration	Monthly when the reactor is critical and prior to return of criticality when a period of subcriticality extends the test beyond 1 month
3.	Control Rod Drop	a.	Verify that all rods move from full out to full in, in less than 2.44 seconds	At each refueling shutdown
4.	(Deleted)			
5.	Pressurizer Safety Valves	a.	Pressure Setpoint	At each refueling shutdown
6.	Main Steam Safety Valves	a.	Pressure Setpoint	At each refueling shutdown
7.	Main Steam Power Operated Relief Valves	a.	Test for Operability	At each refueling shutdown
8.	Trisodium Phosphate Additive	a.	Check for system availability as delineated in Technical Specification 4.2	At each refueling shutdown
9.	Hydrazine Tank Water Samples	a.	Hydrazine concentration	Once every six months when the reactor is critical and prior to return of criticality when a period of subcriticality extends the test interval beyond six months
10.	Transfer Switch No. 7	a.	Verify that the fuse block for breaker 8-1181 to MCC 1 is removed	Monthly, when the reactor is critical and prior to returning reactor to critical when period of subcriticality extended the test interval beyond one month

TABLE 4.1.2 (continued)

		Check	Frequency
11.	MOV-LCV-1100 C Transfer Switch	a. Verify that the fuse block for either breaker 8-1198 to MCC 1 or breaker 42-12A76 to MCC 2A is removed.	Same as Item 10 above
12.	Emergency Siren Transfer Switch	a. Verify that the fuse block for either breaker 8-1145 to MCC 1 or breaker 8-1293A to MCC 2 is removed	Same as Item 10 above
13.	Communication Power Panel Transfer Switch	a. Verify that the fuse block for either breaker 8-1195 to MCC 1 or breaker 8-1293B to MCC 2 is removed	Same as Item 10 above
14a.	Spent Fuel Pool Water Level	Verify water level per Technical Specification 3.8	a. Once every seven days when spent fuel is being stored in the pool.
b.	Refueling Pool Water Level		b. Within two hours prior to start of and at least once per 24 hours thereafter during movement of fuel assemblies or RCC's.
15.	Reactor Coolant Loops/ Residual Heat Removal Loops	a. Per Technical Specifications 3.1.2.C and 3.1.2.D, in Mode 1 and Mode 2 and in Mode 3 with reactor trip breakers closed, verify that all required reactor coolant loops are in operation and circulating reactor coolant.	a. Once per 12 hours
		b. Per Technical Specification 3.1.2.E, in Mode 3 with the reactor trip breakers open, verify	

TABLE 4.1.2 (continued)

Check	Frequency
1. At least two required reactor coolant pumps are operable with correct breaker alignments and indicated power availability.	1. Once per 7 days
2. The steam generators associated with the two required reactor coolant pumps are operable with secondary side water level ≥ 256 inches of narrow range on cold calibrated scale.	2. Once per 12 hours
3. At least one reactor coolant loop is in operation and circulating reactor coolant.	3. Once per 12 hours
c. Per Technical Specification 3.1.2.F, in Mode 4 verify	
1. At least two required (RC or RHR) pumps are operable with correct breaker alignments and indicated power availability.	1. Once per 7 days
2. The required steam generators are operable with secondary side water level ≥ 256 inches of narrow range on cold calibrated scale.	2. Once per 12 hours
3. At least one reactor coolant loop/RHR train is in operation and circulating reactor coolant.	3. Once per 12 hours
d. Per Technical Specifications 3.1.2.G and 3.1.2.H, in Mode 5 verify, as applicable:	

		Check	Frequency
		1. At least one RHR train is in operation and circulating reactor coolant.	1. Once per 12 hours
		2. When required, one additional RHR train is operable with correct pump breaker alignments and indicated power availability.	2. Once per 7 days
		3. When required, the secondary side water level of at least two steam generators is ≥ 256 inches of narrow range on cold calibrated scale.	3. Once per 12 hours
	e.	Per Technical Specification 3.8.A.3, in Mode 6, with water level in refueling pool greater than elevation 40 feet 3 inches, verify that at least one method of decay heat removal is in operation and circulating reactor coolant at a flow rate of at least 400 gpm.	e. Once per 12 hours
	f.	Per Technical Specification 3.8.A.4, in Mode 6, with water level in refueling pool less than elevation 40 feet 3 inches, verify	
		1. At least one decay heat removal method is in operation and circulating reactor coolant.	1. Once per 12 hours
		2. One additional decay heat removal method is operable with correct pump breaker alignments and indicated power availability.	2. Once per 7 days
16. RWST Contained Water Volume	a.	Verify volume ≥ 50 ft. plant elevation	a. Monthly when the reactor is critical and prior to return of criticality when a period of subcriticality extends the surveillance beyond 1 month

ATTACHMENT 3

ACCIDENT ANALYSES